NUCLEAR REACTOR PHYSICS

Weston M. Stacey
Georgia Institute of Technology

A WILEY-INTERSCIENCE PUBLICATION
JOHN WILEY & SONS, INC.
New York • Chichester • Weinheim • Brisbane • Singapore • Toronto

This book is printed on acid-free paper. ♾

Published simultaneously in Canada.

For ordering and customer service, call 1-800-CALL-WILEY.

Library of Congress Cataloging-in-Publication Data:

Stacey, Weston M.
Nuclear reactor physics / Weston M. Stacey.
p. cm.
Includes index.
ISBN 0-471-39127-1 (cloth : alk. paper)
1. Nuclear engineering. 2. Nuclear reactors. 3. Nuclear fission. 4. Nuclear physics. I. Title.

TK9145.S73 2001
621.48'31—dc21 00-043578

Printed in the United States of America.

10 9 8 7 6 5 4 3 2 1

To Penny, Helen, Billy, and Lucia

Criticality Safety Analysis

At various stages of the enrichment, fabrication, and transportation procedures prior to loading the fuel into the reactor, and at various stages of the temporary storage, processing, transportation, and permanent storage procedures for spent nuclear fuel, the nuclear fuel is distributed within a variety of configurations. Examples of such configurations are spent fuel assemblies stored in a swimming pool (to provide for decay heat removal) at the reactor site and barrels of processed fuel in liquid form arrayed on storage racks. Criticality safety requires a rigorous fuel management system to insure that the fuel inventories of each storage element is known and that the various configurations are well subcritical under all normal and conceivable off-normal conditions. Criticality calculations of the type discussed for the case when the fuel is loaded into the reactor must also be performed for these various ex-reactor configurations. While diffusion theory and the methodology discussed in previous chapters may suffice for certain of these configurations, the more rigorous transport methods of Chapter 9 are generally required for criticality safety analyses.

7.11 INTERACTION OF REACTOR PHYSICS AND REACTOR THERMAL HYDRAULICS

Power Distribution

More than 90% of the recoverable energy released in fission is in the form of kinetic energy of fission products and electrons, which is deposited in the fuel within millimeters of the site of the fission event, and somewhat less than 10% of the energy is in the form of energetic neutrons and gamma rays, which are deposited within about 10 cm around the fission site. Thus the heat deposition distribution is approximately the same as the fission rate distribution:

$$q'''(r) \approx \text{const.} \times \Sigma_f(r)\phi(r) \tag{7.1}$$

The requirement to remove this heat without violating constraints on maximum allowable values of materials temperature, heat flux from the fuel into the coolant, and so on, places limits on allowable neutron flux peaking factors, fuel element dimensions, coolant distribution, and so on. The neutron flux distribution affects the temperature in the fuel and coolant/moderator, the temperature of the fuel affects the fuel resonance cross section, and the temperature of the coolant/moderator affects the moderating power, both of which in turn affect the neutron flux distribution.

An increase in the local resonance absorption in the fuel when the local fuel temperature increases results because of the Doppler broadening of the resonances. This increase in local fuel absorption cross section will generally reduce the number of neutrons that reach thermal locally in LWRs, which will tend to reduce the local fission rate and compensate the original increase in fuel temperature. The increase in local fuel resonance absorption makes the fuel compete more effectively

for local neutrons, which tends to make other nearby absorbers somewhat less effective (e.g., reduces the worth of nearby control rods).

The effect of coolant temperature on neutron moderation is also important. In most LWR cores, a local decrease in water density resulting from an increase in water temperature will cause a decrease in neutron moderation, which in turn causes a decrease in local power deposition. As the coolant passes up through the core, the cumulative heat input from the fuel elements causes the axial temperature distribution to increase with height; conversely, the axial density distribution decreases with height. This produces a power distribution peaked toward the bottom of the core, which is pronounced in BWRs, for which progressive coolant voiding occurs in the upper part of the core. Control rods are inserted from the bottom in BWRs to maximize rod worth and to avoid exacerbating this peaking in the axial neutron flux at the bottom of the core. The shift toward a harder spectrum associated with a local Na density decrease in a fast reactor results in an increase in local η, which increases the local heating. The coupling between reactor physics and thermal hydraulics is much weaker in gas-cooled reactors, in which the moderator is separate from the coolant.

Temperature Reactivity Effects

The general reactivity effects associated with changes in fuel, coolant/moderator, and structural temperatures and their effect on the reactor dynamics were discussed in Sections 5.7 to 5.12. The interaction of thermal-hydraulics and reactor physics phenomena to produce positive reactivity in the Three Mile Island and Chernobyl accidents is discussed in Section 8.4. The overall reactivity effect depends on the local changes in temperature and density in each zone of the reactor and the local neutron flux, weighted by the relative importance of these local reactivity contributions and summed over the reactor. The thermal-hydraulics characteristics of a reactor affect not only the local temperature and density changes in response to a change in the neutron flux distribution and magnitude, but also affect changes in the neutron flux distribution and magnitude in response to changes in local temperature and density.

Coupled Reactor Physics and Thermal-Hydraulics Calculations

It is clear from the discussion above that the power distribution and effective multiplication constant in a nuclear reactor depends not only on the distribution of material (fuel, coolant, structure, control) within a reactor core, but also on the temperature and density distribution within a reactor core. In the design process, it is necessary to determine a self-consistent material and temperature–density distribution that makes the reactor critical at operating conditions without violating thermal-hydraulics limits. The problem is further complicated by fuel depletion, which changes the materials in the fuel during the course of time; the distributions of materials and temperature–density must make the reactor critical over its entire lifetime without violating thermal-hydraulics limits. This is normally accomplished

by trial and error, iterating between static neutron flux and thermal-hydraulics calculations until a self-consistent solution is found which can be made critical by adjusting control poison levels and which satisfies thermal-hydraulics and safety limits over the projected core lifetime.

Once the design is fixed, it is necessary to analyze a number of operational and off-normal transients to ensure that the reactor will operate without violation of thermal-hydraulics limits under normal conditions and that it will operate safely under off-normal conditions. The transient analyses codes usually solve for the neutron power amplitude and distribution and the corresponding temperature and density distributions, in some approximation.

REFERENCES

1. "European Pressurized Water Reactor," *Nucl. Eng. Des.*, *187*, 1–142 (1999).
2. D. C. Wade and R. N. Hill, "The Design Rationale of the IFR," *Prog. Nucl. Energy, 31*, 13 (1997); E. L. Gluekler, "U.S. Advanced Liquid Metal Reactor (ALMR)," *Prog. Nucl. Energy, 31*, 43 (1997).
3. R. A. Knief, *Nuclear Engineering*, Taylor & Francis, Washington, DC (1992), Chaps. 8–12.
4. J. G. Collier and G. F. Hewitt, *Introduction to Nuclear Power*, Hemisphere Publishing, Washington, DC (1987), Chaps. 2 and 3.
5. P. J. Turinsky, Thermal Reactor Calculations, in Y. Ronen, ed., *CRC Handbook of Nuclear Reactor Calculations*, CRC Press, Boca Raton, FL (1986).
6. M. Salvatores, "Fast Reactor Calculations," in Y. Ronen, ed., *CRC Handbook of Nuclear Reactor Calculations*, CRC Press, Boca Raton, FL (1986).
7. R. H. Simon and G. J. Schlueter, "From High-Temperature Gas-Cooled Rectors to Gas-Cooled Fast Breeder Reactors," *Nucl. Eng. Des.*, *4*, 1195 (1974).
8. *Report on the Accident at the Chernobyl Nuclear Power Station*, NUREG-1250, U.S. Nuclear Regulatory Commission, Washington, DC (1987).